Materials Reliability Program (MRP) Overview

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Technical Exchange Meeting on Materials, NRC
June 3-5, 2014

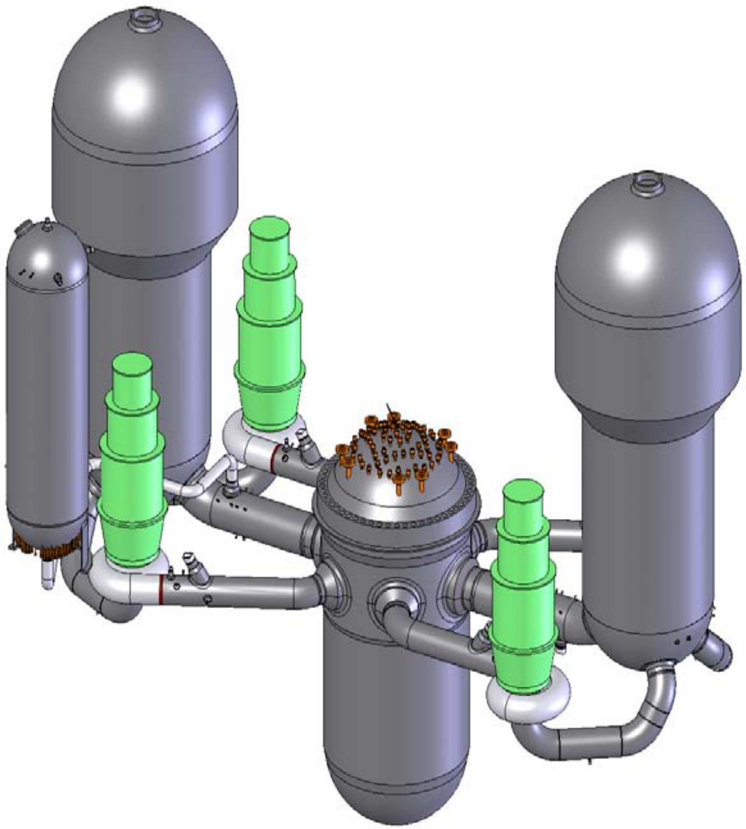
MRP Overview - Outline

- Background and Objective
- MRP Membership and Organization
- 2013/14 PWR Materials Research Gaps
- MRP Guidelines
- 2013 MRP Key Deliverables
- Research Area Examples
- MRP and U.S. NRC Research Cooperation
- Contact Information

MRP Background & Objective

- PWR specific materials issues in the late 1990s led to the formation of the EPRI Materials Reliability Program (MRP) within the Nuclear Sector

- The objective of the MRP is to resolve existing and emerging PWR materials performance, safety, reliability, operational and regulatory issues



EPRI ELECTRIC POWER RESEARCH INSTITUTE

MRP Membership

American Electric Power
Ameren UE
APS
Dominion
Duke Energy
EDF
Entergy
Exelon
FirstEnergy
FPL
The Kansai Electric Power Co., Inc.
KHNP
NMC
OPPD Omaha Public Power District
Pacific Gas and Electric Company
PSEG
Rolls-Royce
SCANA
Southern Company
STP Nuclear Operating Company
TVA
Luminant
Wolf Creek Generating Station
Asociación Española de la Industria Eléctrica UNESA
Yonden
Taiwan power company
Hokkaido Electric Power Co., Inc.

New members & participants in 2013:
- ***EDF Energy***
- ***Vattenfall***
- ***IHI***

2014 MRP Organization

Materials Reliability Program (IC)
Rudell, Constellation
IC Vice Chairman-TBD
Demma, EPRI

Regulatory Interface
Richter, NEI
Dyle, EPRI

Materials Review Visits
Crane, INPO

Technical Advisory Group
IC Vice Chairman-TBD
Demma, EPRI

Technical Support TAC
Hoehn II, Ameren
Childress, Duke Energy
McDevitt, EPRI

Inspection TAC
Fernandez, APS
Cordes, Southern Nuclear
Swain, EPRI

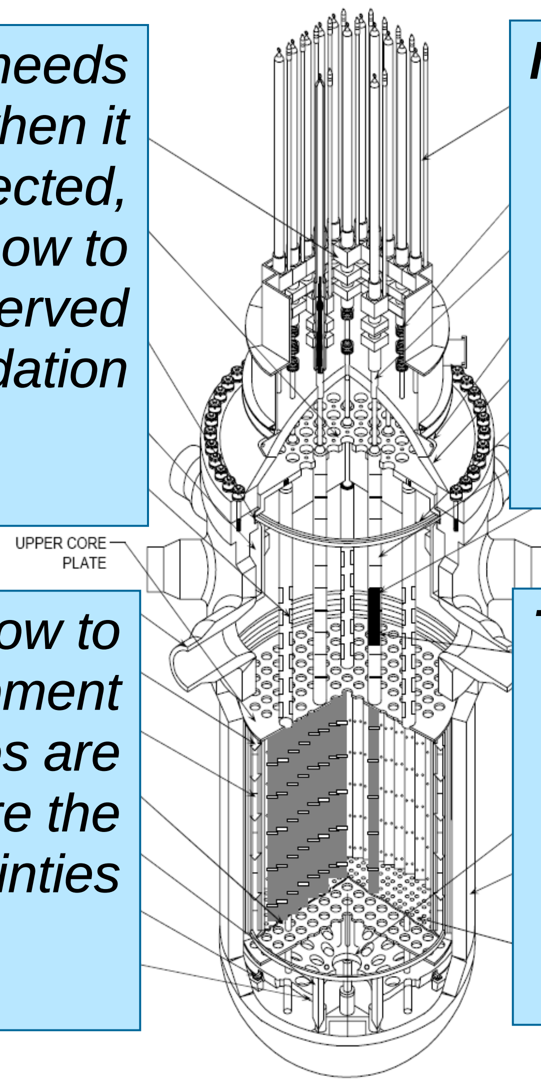
Mitigation/Testing TAC
Gobell, Entergy
Efsing, Ringhals
Smith, EPRI

Assessment TAC
Sims, Entergy
DeBoo, Exelon
Amberge, EPRI
Harrington, EPRI

MRP Technical Advisory Committees

***Assessment** -- What needs to be inspected, when it needs to be inspected, inspection options, how to disposition observed degradation*

***Mitigation and Testing** -- How can degradation be prevented or reduced, irradiated and non-irradiated material testing*



***Inspection** -- How to inspect, what equipment and techniques are available, what are the associated uncertainties*

***Technical Support** – Fatigue and reactor pressure vessel integrity, review and maintain guidelines, compile inspection results*

2013 PWR IMT High Priority Gaps

ID	Gap Description
P-AS-02	Environmental Effects on Fatigue Life: Pressure Boundary Components
P-AS-09	SCC of Stainless Steels Exposed to Primary Water
P-AS-11	PWSCC Crack Growth Rates for Alloys 600, 82, and 182
P-AS-12	PWSCC Factors of Improvement for Alloys 690, 52, and 152
P-AS-13a	Thermal & Irradiation Embrittlement Synergistic Effects on CASS
P-AS-13b	Thermal & Irradiation Embrittlement Synergistic Effects on SS Welds
P-AS-14a	IASCC Characterization: Generic Data Needs
P-AS-14b	IASCC Characterization: Baffle Bolting
P-AS-17	Flow-Induced Vibration and Wear of Reactor Internals
P-AS-19	PWSCC Management for Ni-Alloy Reactor Internals
P-AS-27	Alternative ASME Section XI Appendix G Methodology
P-AS-28	Neutron Embrittlement of Nozzle Forgings and Upper Shell Course
P-AS-38	Fluence Impact on Stainless Steel Mechanical Properties (Fracture Toughness, Tensile Strength)
P-AS-46	CASS Piping Component Thermal Aging Embrittlement & Long-Term Integrity Assess.
P-I&E-03	NDE Technology for J-Groove Weld Locations
P-I&E-12	NDE Technology for Examination of CASS
P-I&E-21	Reactor Internals Generic Acceptance Criteria
P-RG-06	NDE Qualification for Reactor Internals Inspection (VT Evaluation)
P-RG-09	Pipe Rupture Probability Re-Assessment (xLPR)

ID	Gap Description
P-AS-22	Steam Generator Tubes & Internals Wear & High-Cycle Fatigue
P-AS-24	Denting & SCC in Steam Generator Top of Tubesheet (TTS) Region
P-AS-26	Steam Generator Tube Damage due to Loose Parts or Foreign Objects
P-AS-30	ODSCC of Thermally Treated Alloy 600 Steam Generator Tubing
P-AS-31	Safety Significance of Cracks in Steam Generator Divider Plate
P-AS-35	Steam Generator Sludge Deposits and Scale Buildup
P-I&E-15	Steam Generator Tubing Eddy Current Technology Improvements
P-I&E-16	NDE - Tools for Steam Generator Tubing Integrity Assessments
P-I&E-18	Steam Generator Tube Eddy Current Data Analysis Software Improvements
P-I&E-20	Steam Generator Foreign Object Detection and Evaluation Improvements

2013 PWR IMT Medium Priority Gaps

ID	Gap Description
P-AS-04	Neutron Embrittlement of Reactor Pressure Vessel Steels
P-AS-05	Fluence Spectra Effects on Low-Alloy Steel RPV Materials
P-AS-15	Void Swelling of Stainless Steels
P-AS-16	Fatigue Environmental Effects in Reactor Internals
P-AS-36	Outstanding Issues Associated with Thermal Fatigue of ASME Class 1 Piping
P-AS-37	80-Year Reactor Vessel Material Surveillance Program Management
P-AS-45	Equivalent Margin Analysis
P-DM-09	Environmental Effects on Fracture Resistance
P-I&E-08	NDE Technology for Detection and Characterization of Baffle & Former Assembly IASCC
P-I&E-11	NDE Accessibility Evaluation for Reactor Internals
P-I&E-25	NDE of Bottom Mounted Nozzle Penetrations
P-MT-02	PWSCC Mitigation via Surface Treatment Stress Improvement (Peening)
P-MT-09	PWSCC Mitigation via Chemical Surface Treatments
P-RG-05	ASME Section XI, Appendix VIII Flaw Sizing Criteria
P-RG-11	Replacement Component Fitness for Service Acceptance and Acceptance by UT in Lieu of RT
P-RR-03	Welding Processes for Repair of Irradiated Material
P-RR-04	Improved Weldability of Ni Base Alloy Weld Metal
P-RR-06	Repair Guidelines for Reactor Internals
P-RR-08	Alternate Materials for Reactor Internals Repair / Replacement (Esp. Bolting)

ID	Gap Description
P-AS-20	PWSCC of Thermally Treated Alloy 600 Steam Generator Tubing
P-AS-25	Steam Generator Flow-Accelerated Corrosion Assessment
P-AS-44	Steam Generator Channel Head Wastage
P-I&E-13	NDE Capability for Sizing Steam Generator Tubing ODSCC Indications
P-I&E-24	NDE of Steam Generator Channel Head Material
P-MT-04	Steam Generator Tubing ODSCC Mitigation via Water Chemistry Technologies
P-MT-07	Steam Generator Startup Chemistry Excursions after SG Replacement

ID	Gap Description
P-MT-01	PWSCC Mitigation via Water Chemistry Controls (Zn/H2)
P-RG-10	Management of License Renewal Issues

2013 PWR IMT Low Priority Gaps

ID	Gap Description
P-AS-06	Pressurized Thermal Shock Re-Evaluation
P-AS-29	High-Cycle Fatigue Potential at RPV Safety Injection and Core Flood Line Locations
P-AS-39	MRP Reactor Internals Aging Management Program 80-Year Evaluation
P-AS-40	Low Temperature Crack Propagation (LTCP) Assessment
P-DM-10	Thermal Embrittlement of Low-Alloy Pressure Vessel Steels
P-DM-11	SCC (and Thermal Aging) of CASS Pressure Boundary Components
P-DM-12	Increased Fastener SCC Susceptibility due to Long-Term Aging
P-DM-13	Long-Term SCC Susceptibility (Late Life SCC Initiation)
P-DM-14	Long-Term Stability of Surface Stress Improvement Mitigations
P-DM-15	Thermal Embrittlement of Martensitic Stainless Steels
P-I&E-05	I&E Guidance for Alloy 600 “Orphan” Locations
P-I&E-19	NDE Technology for Implementation of Section XI Radiography
P-RG-13	Management of Subsequent License Renewal Issues
P-RR-09	Repair / Replacement Guidance for Thermal Fatigue of ASME Class 1 Piping
P-RR-10	Alternative DM Weld Repair Solutions

ID	Gap Description
P-AS-32	Steam Generator Safety Significance Evaluation for Non-Tubing / Non-Divider Plate Alloy 600 Components
P-AS-41	ODSCC of Nuclear Grade Alloy 800 Steam Generator Tubing and Sleeves
P-AS-42	ODSCC of Thermally Treated Alloy 690 Steam Generator Tubing
P-DM-16	Thermal Embrittlement of Martensitic Stainless Steels (SG Tube Support Plates)
P-MT-10	Guidance for Extended Layup of SGs and BOP Systems
P-RG-08	Steam Generator Eddy Current Noise Measurement & Monitoring
P-RG-12	Steam Generator Improved Tubing Leak Rate Modeling
P-RR-05	Steam Generator Thermally Treated Tubing SCC Alternate Repair Criteria

2013 PWR IMT Gaps: Major Changes from 2010 to 2013

New Gaps (10 Total)

- P-AS-44: SG Channel Head Wastage
- P-AS-45: Equivalent Margin Analyses
- P-AS-46: CASS Piping Thermal Aging Embrittlement & Long-Term Integrity Assess.
- P-I&E-24: NDE of SG Channel Head Material
- P-I&E-25: NDE of Bottom Mounted Nozzle Penetrations
- P-MT-10: Guidance for Extended Layup of SGs and BOP Systems
- P-RR-10: Alternative DM Weld Repair Solutions
- P-RG-11: Replacement Component Fitness for Service
- P-RG-12: SG Improved Tubing Leak Rate (Regulatory)
- P-RG-13: Management of Subsequent License Renewal Issues

Closed Gaps (8 Total)

- P-AS-01: Boric Acid Corrosion in Annulus Regions
- P-AS-34: SG Improved Tubing Leak Rate Modeling
- P-AS-43: PWSCC of Alloy 690TT SG Tubing
- P-I&E-02: NDE Qualification Program for RPV Upper Head Penetrations
- P-I&E-07: NDE Qualification for Reactor Internals Bolting
- P-I&E-14: NDE Capability for Examination of Hidden Welds
- P-I&E-22: Appendix VIII Compliance
- P-I&E-23: NDE Technology for Socket Welded Piping Configurations

Split Gaps

- P-AS-13: Split into 2 gaps - address CASS and SS weld thermal aging separately
- P-AS-14: Split into 2 gaps - address IASCC of baffle bolting & generic IASCC data needs separately

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MRP Documents with Mandatory and Needed Elements Governed by the Materials Initiative

Doc Number (EPRI PID)	Rev	Document Title	Date	Implementation Level	Comments
MRP-126 (1009561)	0	Generic Guidance for an Alloy 600 Management Plan	Nov 2004	Mandatory	
MRP-146 (1022564)	1	Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines	Jun 2011	Needed	
MRP-146S (1018330)	0	Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines – Supplemental Guidance	Jan 2009	Needed	
MRP-227-A (1022863)	A	MRP 227-A, Pressurized Water Reactors Internals Inspection and Evaluation Guidelines	Dec 2011	Mandatory	
MRP 2014-006	0	MRP-227-A Interim Guidance Modification to inspection requirements of tables 4-3 and 5-3 for Westinghouse Control Rod Guide Tube Assemblies	Feb 2014	Needed	MRP Letter

Documents Incorporated Within (i.e., issued prior to the initiative) or Under the Materials Initiative (i.e., issued since the initiative)

MRP Documents with Mandatory and Needed Elements Governed by the Materials Initiative

Doc Number (EPRI PID)	Rev	Document Title	Date	Implementation Level	Comments
MRP-228 (1025147)	1	MRP-228 Inspection Standard for PWR Internals	Dec 2012	Needed	
MRP 2013-023	0	MRP-228 Interim Guidance Reactor Internal Baffle-Former Bolting Ultrasonic Examinations	Oct 2013	Needed	Letter
MRP-139 (1015009) MRP 2010-046	1	MRP-139, Primary System Piping Butt Welds Inspection and Evaluation Guideline, and Interim Guidance Letter MRP 2010-046	Dec 2008	N/A	Superseded by requirements of Code Case N-770 as amended by 50.55a Listed for historical purposes only. MRP maintains the technical basis.
MRP-326 (1022871) MRP 2012-031	0	MRP-326, Coordinated PWR Vessel Surveillance Program (CRVSP) Guidelines, and Interim Guidance Letter MRP 2012-031	Dec 2011	N/A	All required member actions have been completed. Listed for historical purposes only.

Documents Incorporated Within (i.e., issued prior to the initiative) or Under the Materials Initiative (i.e., issued since the initiative)

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2013 Key Deliverables

1. **Evaluation of the Reactor Vessel Beltline Shell Forgings of Operating U.S. PWRs for Quasi-Laminar Indications (MRP-367)**
   - NDE fabrication record reviews and a flaw detection capabilities study resulted in high confidence that quasi-laminar indications observed at Doel 3 and Tihange 2 would have been detected and are not present in US PWRs
   - Bounding structural evaluation demonstrated that reactor vessel safety goals are met through the extended license period in the unlikely event that flaking exists in beltline ring forgings
2. **PWR Bottom Mounted Nozzle Issue Response Handbook (MRP-372)**
   - Compilation of available knowledge and reference information to support BMN contingency planning by utilities, including the development of a site-specific response plan to an emergent BMN issue
3. **Resistance of Alloys 690, 152 and 52 to Primary Water Stress Corrosion Cracking (MRP-237, Rev. 2)**
   - Summarizes the state of knowledge and remaining research gaps on A690 and its welds metals

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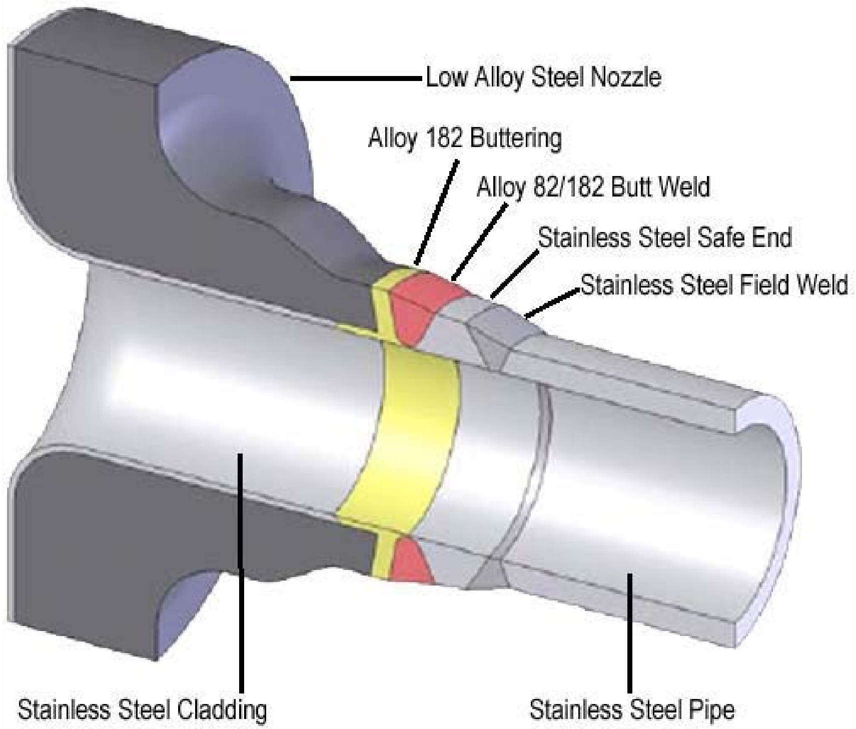
MRP Research Area: Alloy 600

Alloy 600 Management Tools

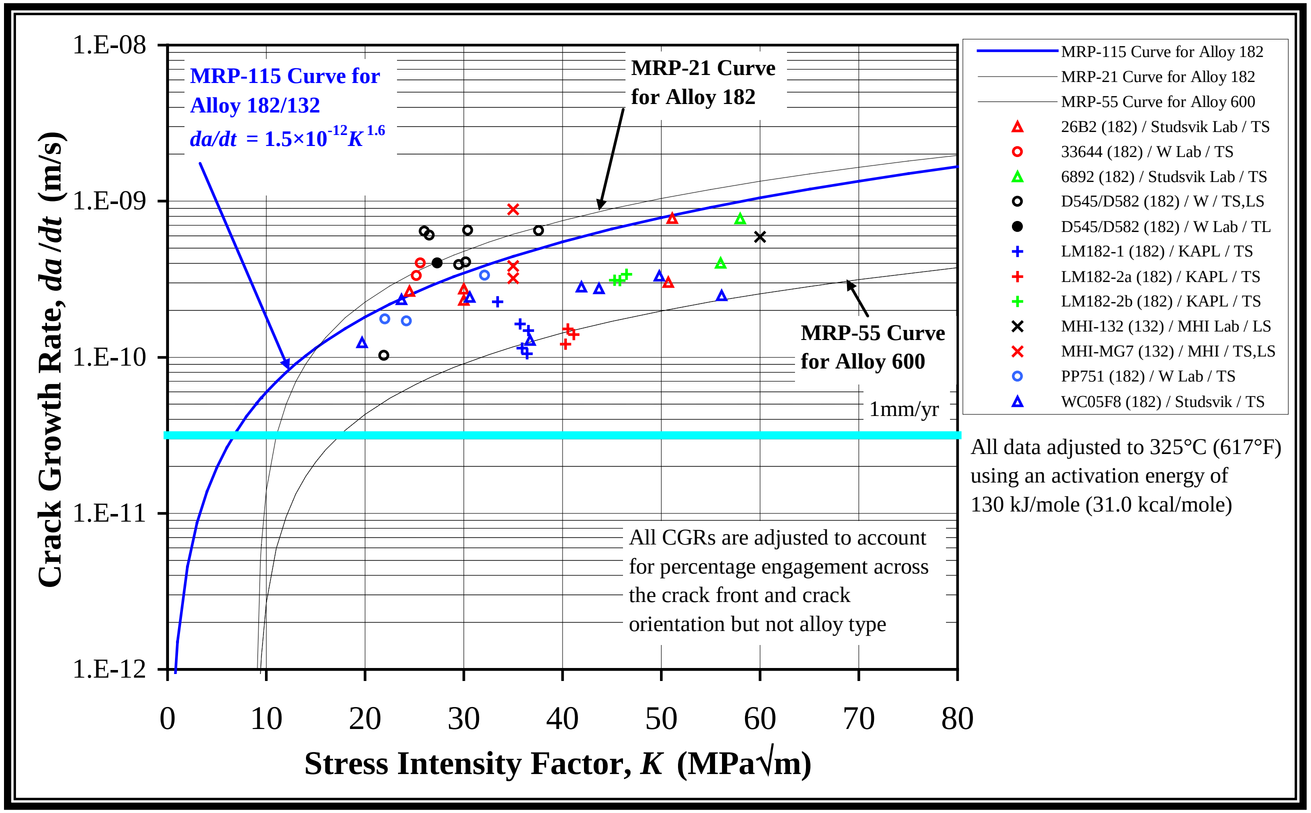
- Alloy 600 Management Program: MRP-126
- RV Upper Head Penetrations: MRP-117
  - *Technical basis for ASME Code Case N-729*
  - *Bases being reassessed to incorporate early occurrence of T_{cold} RV Head PWSCC OE*
- A82/182 DM Buttwelds: MRP-139
  - *Technical bases support ASME CC N-770*
- RV Bottom-Mounted Nozzles: MRP-206
  - *Technical bases support ASME CC N-722 BMN exams*
  - *Bases reassessed to consider EdF BMN PWSCC at Gravelines 1 and Palo Verde 3*
- Alloy 600 Inspection Tracking: MRP-219
  - *Updated annually*



MRP Research Area: Alloy 600 (cont.)

MRP Crack Growth Rate Curves for Alloys 600 and 182/132

- Widely referenced
- Incorporated into ASME Boiler & Pressure Vessel Code, Section XI
- Revision planned to update curves incorporating new data and model



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MRP Research Area: PWSCC Mitigation by Peening

- Readiness, Status and Plans

Light Water Reactors in Japan

- 12+ years of peening OE in PWRs and BWRs
- 23+ PWRs mitigated, in-situ during RFOs
- Laser and Cavitation (WJP) technologies
- Alloy 600 Nozzles, J-Welds and DM Butt-welds

MRP R&D Program Complete

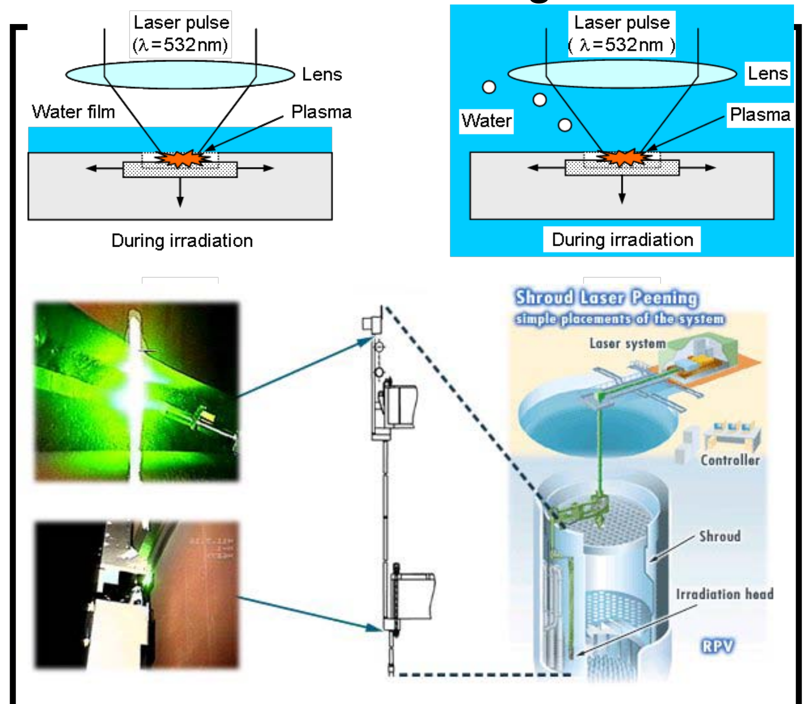
- PWSCC Initiation Testing
- Residual Stress Relaxation (Testing & Modeling)
- Vendor Technical Basis Information

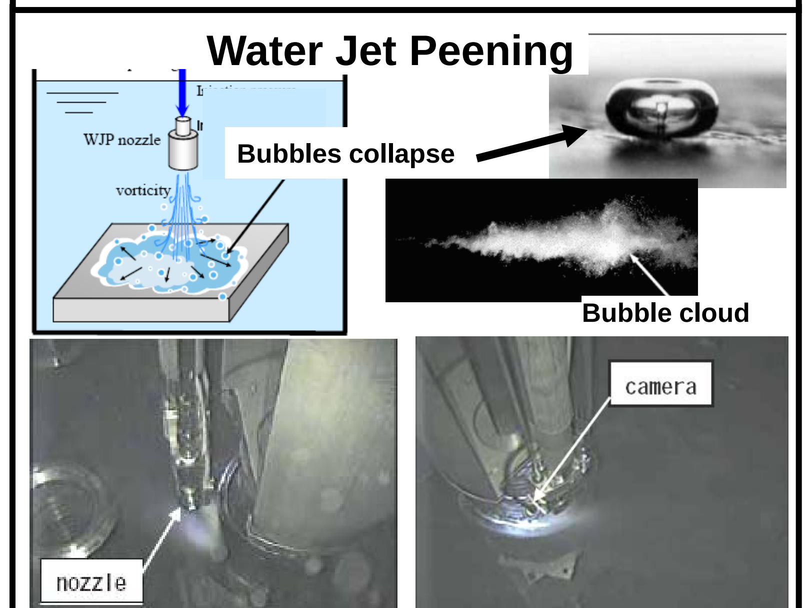
Implementation Documentation Submitted to NRC for SE and for ASME Code Cases

- Technical Basis Document (MRP-267, Rev 1)
- Topical Report for Inspection (MRP-335, Rev 1)

Plans for peening two US PWRs in 2016 have been announced.

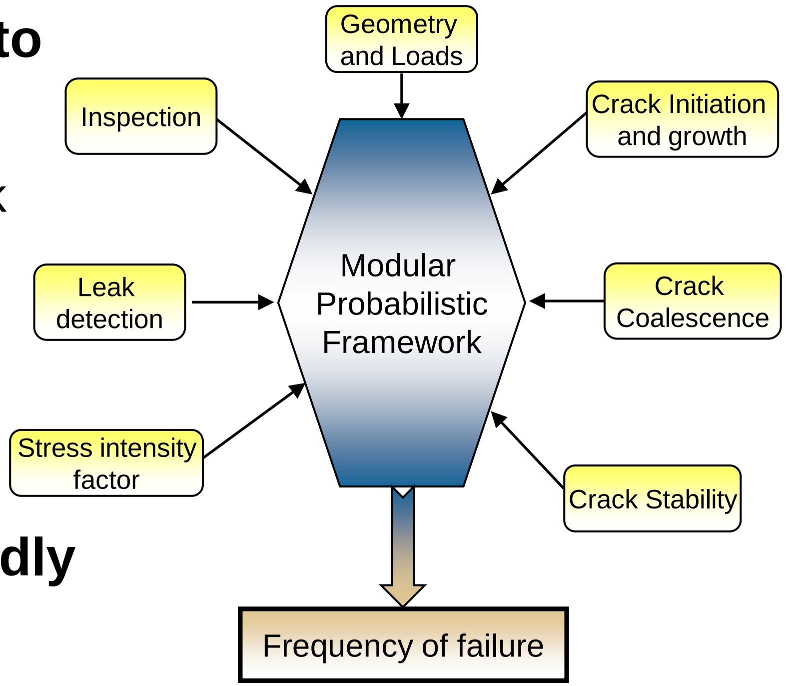
Laser Peening





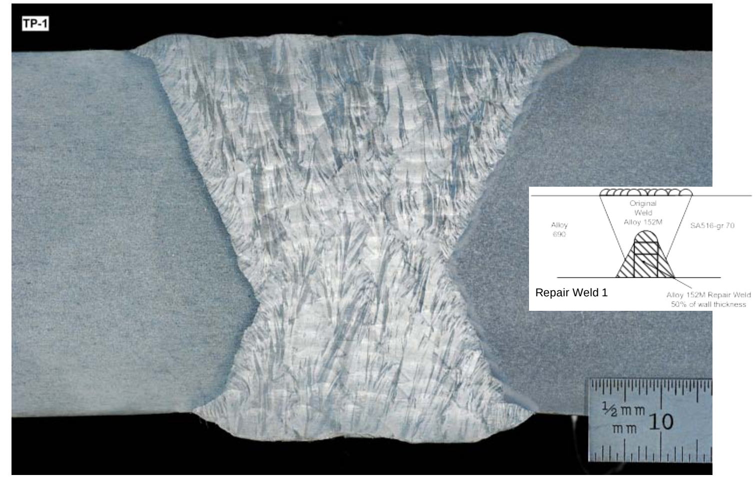
MRP Research Area: Extremely Low Probability of Rupture (xLPR)

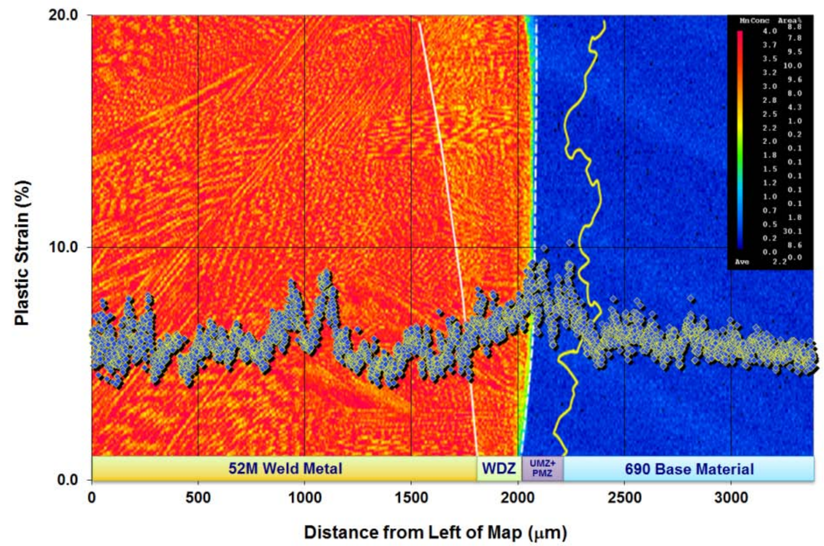
- **Fully cooperative program with NRC Research to develop a probabilistic fracture mechanics approach initially to**
  - Address presence of PWSCC in piping previously approved for leak-before-break
  - Conduct more realistic flaw evaluations
  - Evaluate reduction in risk with mitigation
  - Research prioritization studies
- **Further development could more broadly apply probabilistic tools to**
  - BWRs as well as PWRs
  - Other degradation mechanisms and materials
  - Broader range of components



MRP Research Area: Alloy 690

- Testing and field experience confirm high PWSCC resistance of Alloys 690/52/152
- More extensive work explored or underway to further understand
  - Vulnerability to abnormal microstructure and to welding residual stresses and strains
  - Role of weld imperfections in initiation and growth of PWSCC
- Work substantiates technical basis for optimized inspection of resistant material replacement components such as the reactor pressure vessel head





MRP Research Area: Alloy 690 (cont.)

International Collaboration

EPRI MRP Alloy 690 Project

MHI/IHI

NRC

EDF / MAI

KAERI

Bettis / KAPL

Tohoku University

UNESA

PWSCC Degradation Characterization

Guidelines for A690 Material Procurement

CGR Data for Flaw Disposition Curves

A690/52/152 Macro and Microstructural Mapping and Strain Analysis

PWSCC Resistance of Evolutionary A52 Compositions

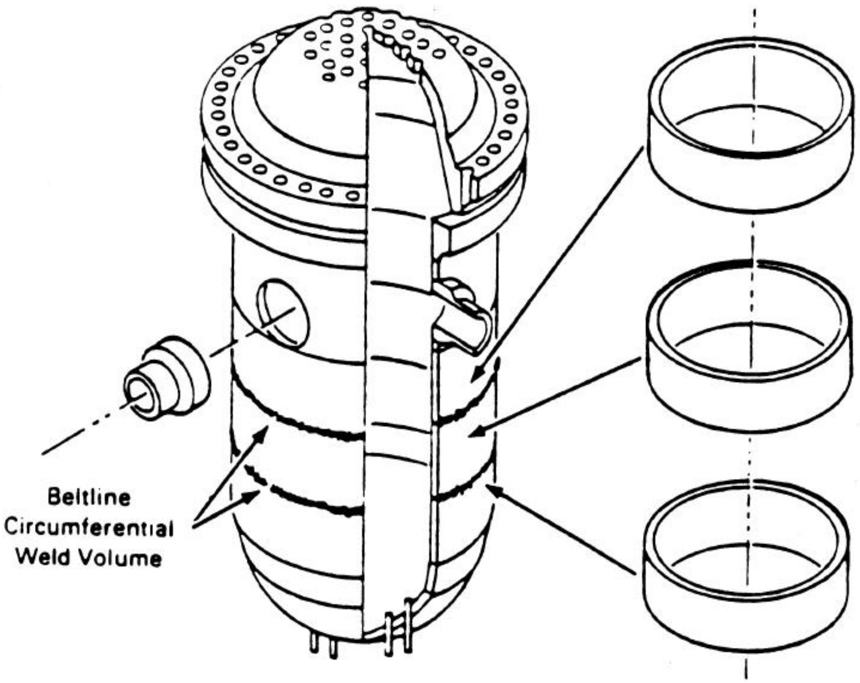
HAZ PWSCC Degradation Characterization

Expert Panel

Projects

MRP Research Area: Reactor Pressure Vessel

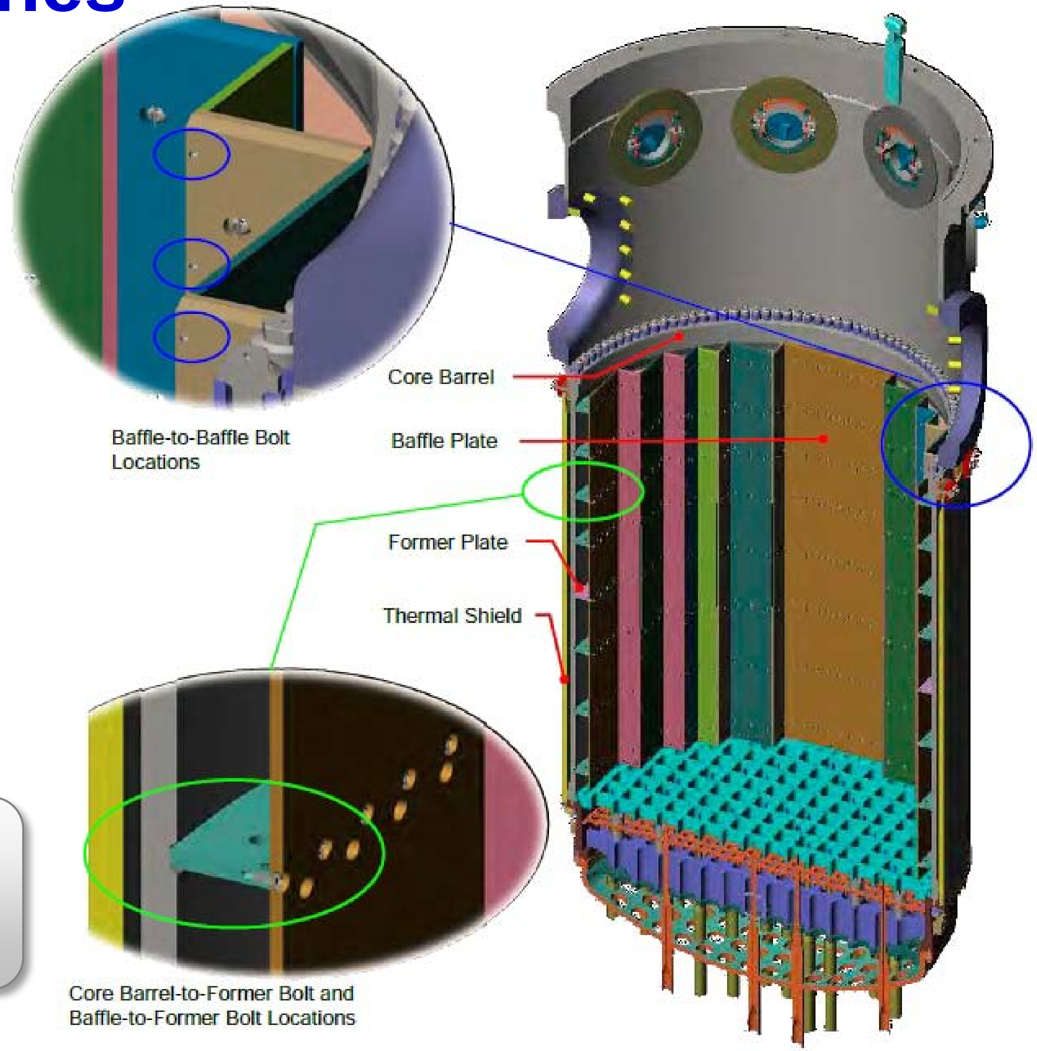
- Extending research to address RPV integrity issues through 80 years
- Generate PWR surveillance data to inform embrittlement trend correlation (ETC) applicable through 80 year fluence
- Degradation modeling
  - Charpy Master Curve ETC
  - Hydrogen flaking
  - Appendix G small flaw issues
- Develop tools and conduct training for RPV embrittlement/ integrity issues



MRP Research Area: PWR Internals Inspection & Evaluation Guidelines

- PWR Internals Inspection and Evaluation Guidelines MRP-227-A (Report 1022863) issued in Jan. 2012
  - Inspection guidance and flaw evaluation methodologies

Experience to date shows limited degradation



Typical internals core barrel assembly for B&W-designed PWRs

MRP Research Area: Irradiated Materials Testing – Zorita Research Project (ZIRP)

Project extracted Zorita Reactor Internals material irradiated under service conditions and is conducting extensive testing to increase understanding of fluence effects on:

- *Mechanical properties*: tensile strength, fracture toughness, crack initiation and growth
- *Microscopic properties*: grain boundary chemistry and size, void formation, and hydrogen and helium production

Jose Cabrera NPP “Zorita”
Westinghouse design
1968 – 2006 (~26 EFPY)

Participants: MRP, U.S. NRC, Tractebel, AXPO, Ringhals, SSM

- Additional in-kind contribution from Japanese utilities/MHI
- BWRVIP, PSCR, and U.S. NRC planning to use additional Zorita materials for other research projects

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EPRI MRP/U.S. NRC - Cooperative Research

- Current
  - A690 PWSCC Crack Growth Testing
  - Extremely Low Probability of Rupture (xLPR)
  - Welding Residual Stress (WRS) FEA Model Validation
  - Irradiation Material Testing: Zorita Internals Research Project (ZIRP)

- Potential
  - A690/600 PWSCC Crack Initiation Testing
  - Reactor Pressure Vessel (RPV) Integrity
  - Irradiation Material Testing: additional Zorita materials

Concluding Remarks

- MRP is focused on the understanding and resolution of materials issues for PWR primary components
- MRP research efforts and guidance have addressed material concerns and contributed to improved safety and reliability of the PWR fleet
- Appropriate prioritization and collaboration can focus materials issue research work to most efficiently and effectively solve or manage material issues arising with age and operating experience

MRP Contact Information

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